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RBG-46974

November 17, 2009

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Subject: Licensee Event Report 50-458 / 09-002-00
River Bend Station – Unit 1
Docket No. 50-458
License No. NPF-47

File No. G9.5

RBF1-09-0150

Dear Sir or Madam:

In accordance with 10CFR50.73, enclosed is the subject Licensee Event Report.
This document contains no commitments.

Sincerely,


David N. Lorfing
Manager – Licensing

DNL/dhw
Enclosure

IE22
NRK

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cc: U. S. Nuclear Regulatory Commission
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INPO Records Center
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LICENSEE EVENT REPORT (LER)

(See reverse for required number of
digits/characters for each block)

Estimated burden per response to comply with this mandatory collection request: 50 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the Records and FOIA/Privacy Service Branch (T-5 F52), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to infocollects@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202, (3150-0104), Office of Management and Budget, Washington, DC 20503. If a means used to impose an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

1. FACILITY NAME River Bend Station – Unit 1	2. DOCKET NUMBER 05000-458	3. PAGE 1 of 4
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4. TITLE Unplanned Manual Reactor Scram Following Trip of Both Reactor Recirculation Pumps

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO.	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
09	20	2009	2009	- 002 -	00	11	17	2009		05000
									FACILITY NAME	DOCKET NUMBER
										05000

9. OPERATING MODE 1	11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR§: (Check all that apply)									
	<input type="checkbox"/> 20.2201(b)	<input type="checkbox"/> 20.2203(a)(3)(i)	<input type="checkbox"/> 50.73(a)(2)(i)(C)	<input type="checkbox"/> 50.73(a)(2)(vii)						
10. POWER LEVEL 23	<input type="checkbox"/> 20.2201(d)	<input type="checkbox"/> 20.2203(a)(3)(ii)	<input type="checkbox"/> 50.73(a)(2)(ii)(A)	<input type="checkbox"/> 50.73(a)(2)(viii)(A)						
	<input type="checkbox"/> 20.2203(a)(1)	<input type="checkbox"/> 20.2203(a)(4)	<input type="checkbox"/> 50.73(a)(2)(ii)(B)	<input type="checkbox"/> 50.73(a)(2)(vii)(B)						
	<input type="checkbox"/> 20.2203(a)(2)(i)	<input type="checkbox"/> 50.36(c)(1)(i)(A)	<input type="checkbox"/> 50.73(a)(2)(iii)	<input type="checkbox"/> 50.73(a)(2)(ix)(A)						
	<input type="checkbox"/> 20.2203(a)(2)(ii)	<input type="checkbox"/> 50.36(c)(1)(ii)(A)	<input checked="" type="checkbox"/> 50.73(a)(2)(iv)(A)	<input type="checkbox"/> 50.73(a)(2)(x)						
	<input type="checkbox"/> 20.2203(a)(2)(iii)	<input type="checkbox"/> 50.36(c)(2)	<input type="checkbox"/> 50.73(a)(2)(v)(A)	<input type="checkbox"/> 73.71(a)(4)						
	<input type="checkbox"/> 20.2203(a)(2)(iv)	<input type="checkbox"/> 50.46(a)(3)(ii)	<input type="checkbox"/> 50.73(a)(2)(v)(B)	<input type="checkbox"/> 73.71(a)(5)						
	<input type="checkbox"/> 20.2203(a)(2)(v)	<input type="checkbox"/> 50.73(a)(2)(i)(A)	<input type="checkbox"/> 50.73(a)(2)(v)(C)	<input type="checkbox"/> OTHER						
	<input type="checkbox"/> 20.2203(a)(2)(vi)	<input type="checkbox"/> 50.73(a)(2)(i)(B)	<input type="checkbox"/> 50.73(a)(2)(v)(D)							
Specify in Abstract below or in NRC Form 366A										

12. LICENSEE CONTACT FOR THIS LER

FACILITY NAME David N. Lorfing, Manager – Licensing	TELEPHONE NUMBER (Include Area Code) 225-381-4157
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13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX
E	ED	BKR	(see text)	yes	B	ED	CBL4	(see text)	yes

14. SUPPLEMENTAL REPORT EXPECTED

☐ YES (If yes, complete 15. EXPECTED SUBMISSION DATE)☒ NO

15. EXPECTED SUBMISSION DATE

MONTH	DAY	YEAR

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)

On September 20, 2009, at 5:47 p.m. CDT, an unplanned manual reactor scram was initiated while reactor power was at approximately 23 percent of rated thermal power. A plant shutdown for a refueling outage was in progress at the time. At approximately 5:33 p.m., with reactor power at 37 percent, the operators had attempted to shift the reactor recirculation pumps from fast speed to slow in accordance with the plant shutdown procedure. RCS pump "A" shifted to slow speed, but approximately nine seconds later, it tripped off. RCS pump "B" did not complete the transfer to slow speed, but tripped off from high speed. Following the trip of the RCS pumps, the thermal-hydraulic conditions in the reactor remained within operational limits of plant Technical Specifications. The response procedures for the trip of both RCS pumps did not specify the actuation of the manual scram. The conservative decision was made to initiate the scram. After the scram signal was initiated, operators stabilized reactor pressure and temperature. All systems responded as designed. No safety systems were out of service at the onset of the event. A controlled reactor cooldown was commenced in preparation for the outage. This event is being reported in accordance with 10CFR50.73(a)(2)(iv)(A) as a condition that resulted in the unplanned manual actuation of the reactor protection system while the reactor was critical.

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REPORTED CONDITION

On September 20, 2009, at 5:47 p.m. CDT, an unplanned manual reactor scram was initiated while reactor power was at approximately 23 percent of rated thermal power. A plant shutdown for a refueling outage was in progress at the time. At approximately 5:33 p.m., with reactor power at 37 percent, the operators had attempted to shift the reactor recirculation (RCS) (AD) pumps (**P**) from fast speed to slow in accordance with the plant shutdown procedure. RCS pump "A" shifted to slow speed, but approximately nine seconds later, it tripped off. RCS pump "B" did not complete the transfer to slow speed, but tripped off from high speed.

Following the trip of the RCS pumps, the thermal-hydraulic conditions in the reactor remained within operational limits of plant Technical Specifications. The response procedures for the trip of both RCS pumps did not specify the actuation of the manual scram. The conservative decision was made to initiate the scram.

After the scram signal was initiated, operators stabilized reactor pressure and temperature. All systems responded as designed. No safety systems were out of service at the onset of the event. A controlled reactor cooldown was commenced in preparation for the outage.

This event is being reported in accordance with 10CFR50.73(a)(2)(iv)(A) as a condition that resulted in the unplanned manual actuation of the reactor protection system while the reactor was critical.

IMMEDIATE ACTIONS

The condition of the control logic and the circuit breakers supplying the RCS pumps was investigated. At approximately 9:56 p.m., RCS pump "B" was successfully started in slow speed.

BACKGROUND

The RCS system comprises two identical loops connected to the reactor pressure vessel (RPV), each containing a two-speed motor-driven pump and a flow-control valve. RCS loop flow is directed through 20 jet pumps inside the RPV as drive flow through the core.

When operating in high speed, the RCS pumps run at 1800 RPM, powered directly from non-safety related 13.8KV switchgear via a step-down transformer that supplies 4160 volts to the pump motors. When operating in slow speed, a low-frequency motor-generator set (LFMG) (**MG**) (EC) for each loop supplies power at 15 Hz to the pump motors, running them at 450 RPM.

A logic circuit controls the actuation and timing of the various circuit breakers needed to operate the RCS pumps. When the pump speed is shifted, the logic automatically actuates the circuit breakers.

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CAUSAL ANALYSIS

Since the "A" pump did shift from high to slow speed (although only briefly), it was determined that each loop had exhibited a different failure mode.

Loop "A"

The trip signal that caused the shutdown of RCS pump "A" originated with the ground fault detection relay for the "A" LFMG. The investigation identified an electrical short to ground in the "C" phase between the inboard primary containment penetration and the RCS pump motor termination box. Visual inspection of the cable (**CBL4**) found that the stress cone at the motor terminal connection had failed. This appeared as a small-diameter hole through the insulation at the edge of the semiconductor. Based on industry experience, this was most likely caused by a slight nick in the insulation where the semi-conductive layer begins. This slight nick would be caused by an installation error due to over-cutting the depth of the semiconductor layer during stress cone construction. The failed stress cone was original equipment installed during plant construction.

Loop "B"

The failure of RCS pump "B" to downshift was traced to a malfunction of the 480 volt circuit breaker (**BKR**) supplying the motor on the "B" LFMG. The breaker was subsequently disassembled, and it was determined that the malfunction was due to inadequate lubrication of the breaker mechanism and closing coil assembly. This breaker was one of a group of 480 volt breakers (General Electric Model "AKR") scheduled for replacement during the refueling outage.

AKR circuit breakers have been previously identified as a group of components needing replacement to improve the reliability of the served components. A replacement plan for these breakers was developed in 2007. That plan prioritized replacement of AKR breakers based on criticality and safety function related to breaker tripping. (As of the date of this event, all AKR breakers supplying safety-related components have been replaced. Of 293 breakers in non-safety related service, 105 have been replaced.) The prioritization plan considered non-safety related breakers with closure functions which are critical to operations. However, there was no mitigation strategy in place to ensure operationally significant breakers with closure functions were maintained until time of replacement.

CORRECTIVE ACTION TO PREVENT RECURRENCE

The stress cone on the "A" RCS pump was repaired. A review of operating experience at RBS did not identify any trend with stress cone failures due to installation errors.

The 480 volt circuit breaker supplying the "B" LFMG set was replaced. Additionally, the station will evaluate the prioritization/mitigation plan for breaker replacement to ensure the appropriate mitigating strategies are in place based upon breaker criticality and closure functions that may result in operational concerns.

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PREVIOUS OCCURRENCE EVALUATION

No previous reactor scrams occurring at River Bend Station within the past five years were attributable to the same root cause as this event.

SAFETY SIGNIFICANCE

A trip of two recirculation pumps is evaluated in the RBS Updated Safety Analysis Report. That evaluation assumes that the plant is operating at 100 percent power at the time of the event. The analysis concludes that there is no change in minimum critical power ratio for the event. Further, the analysis indicates that reactor water level would increase due to decrease in core flow and cause a reactor scram / turbine trip on high reactor water level (Level 8). This event occurred at a much lower thermal power, and did not result in a Level 8 scram / turbine trip. As such, the event response was much milder than the event analyzed in the USAR.

The postulated events which establish reactor thermal limits are much more severe than the reported event, so there was no challenge to fuel integrity. Reactor pressure did not increase during the transient, and therefore, there was no challenge to the reactor vessel integrity. No energy was released to the containment therefore no challenge to the containment integrity. In summary, all three fission product barriers remained unchallenged.

The reactor power / core flow state point terminated outside the exclusion region of the core operating limits report. No thermal-hydraulic oscillations were observed prior to the actuation of the manual scram signal.

(NOTE: Energy Industry Component Identification codes are annotated as (**XX**).)